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PARAMETRIC STUDY OF PERIPHERAL CONTROL EFFECTIVENESS ON LARGE CORE ROCKET REACTORS

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Lewis Research Center
Cleveland, Ohio

CLASSIFICATION CHANGED
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Date Jan. 3, 1973
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PARAMETRIC STUDY OF PERIPHERAL CONTROL EFFECTIVENESS ON LARGE CORE ROCKET REACTORS

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SUMMARY

This report presents the results of a parametric study of large core graphite-uranium matrix nuclear reactors to determine the characteristics of a reflected drum control system. A one-dimensional, multigroup, multiregion diffusion code, written for the IBM 7094, was used for the study. Critical core sizes for various reactor configurations were first determined, such that the static criticality factor $K_{eff} = 1.050$, and then their effective control span worths were calculated. The reactor parameters varied in this study were core void fraction, core moderator to fuel ratio, core height, beryllium reflector (thickness and void), and the composition of the inner support annulus.

The determination of core diameters agrees rather well with the experimental and theoretical data published. The method of calculating peripheral control worth used in this report agrees to within 5 percent of the accepted values of the Kiwi reactor.

This investigation shows that as one increases the core void fraction, while maintaining a constant K_{eff} (i. e., increasing the core diameter) the peripheral control worth declines. Increasing the moderator to fuel ratio does yield an increase in effective control at the large core diameters (~60 in.). Increases in either the beryllium reflector thickness or the core height also increases the effective worth of the control system. The beryllium reflector thickness is the most predominant factor in determining the control worth span.

INTRODUCTION

At present, the primary nuclear reactor being considered for space propulsion applications utilizes a graphite-uranium matrix through which hydrogen is passed, heated to a very high temperature, and then expelled through a nozzle. The reactor power attained from such a system is about a thousand megawatts. Mission studies indicate that higher powered nuclear rocket engines are required. The problems associated with extrapolat-

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ing this nuclear engine concept to larger core sizes and higher power outputs are manifold. One of these problems is that of reactor control.

The present nuclear rocket reactors are controlled by the positioning of rotating absorbing control drums in the outer beryllium reflector. This form of control is considered desirable for the larger, higher power reactors. Reactor geometries favorable for heat transfer, fluid flow, nozzle durability, and pump capabilities may have diameters so large that peripheral control drums may not give adequate control.

The purpose of this report is to indicate the influence of various reactor parameters on the effectiveness of the peripheral control drum system. The reactor parameters studied were core diameter, core void, moderator to fuel ratio, reflector thickness, and void and core height.

A great deal of work, of a classified nature, has been done at both Los Alamos Scientific Laboratory (LASL) and Westinghouse Astronuclear Laboratory to analytically determine the peripheral control span of graphite-uranium matrix reactors, such as Kiwi and NRX (refs. 1 and 2). The technique developed by LASL to determine control drum worth was successful for the Kiwi design and will be adopted for this report.

This study investigates the control trends resulting from a wider variation of reactor parameters than have been previously reported in the literature. The ranges of the major reactor parameters are as follows: core diameters from 30 to 70 inches, core void fractions from 20 percent to 40 percent, moderator to fuel ratios from 100 to 3000, beryllium reflector thicknesses from 4. 5 to 12. 5 inches, and core heights from 52 to 60 inches.

DISCUSSION OF CALCULATIONAL METHODS

A peripheral drum configuration for a nuclear rocket engine can be seen in figure 1. The 12 cylindrical drums located in the beryllium reflector each have a 120° sheath of boron 10 attached to their circumference and running the length of the core. For the course of this study, the drum diameter will always equal the reflector thickness. Rotation of these peripheral drums positions the neutron absorbing boron vanes and gives the necessary means for controlling the reactor. Turning of the drums from the "full-in" position (boron vanes closest to the core) to the "full-out" position (boron vanes furthest from the core) results in a change in the effective multiplication factor. The difference in effective multiplication between these two vane positions is known as drum span or swing worth.

Drum span for the various reactor models in this study was calculated by a two-step process involving the use of a one-dimensional, multigroup, multiregion diffusion code. The initial portion of the calculation consisted of expanding or contracting the core diameter for the various reactor configurations until a K_{eff} equal to 1. 050 was established.

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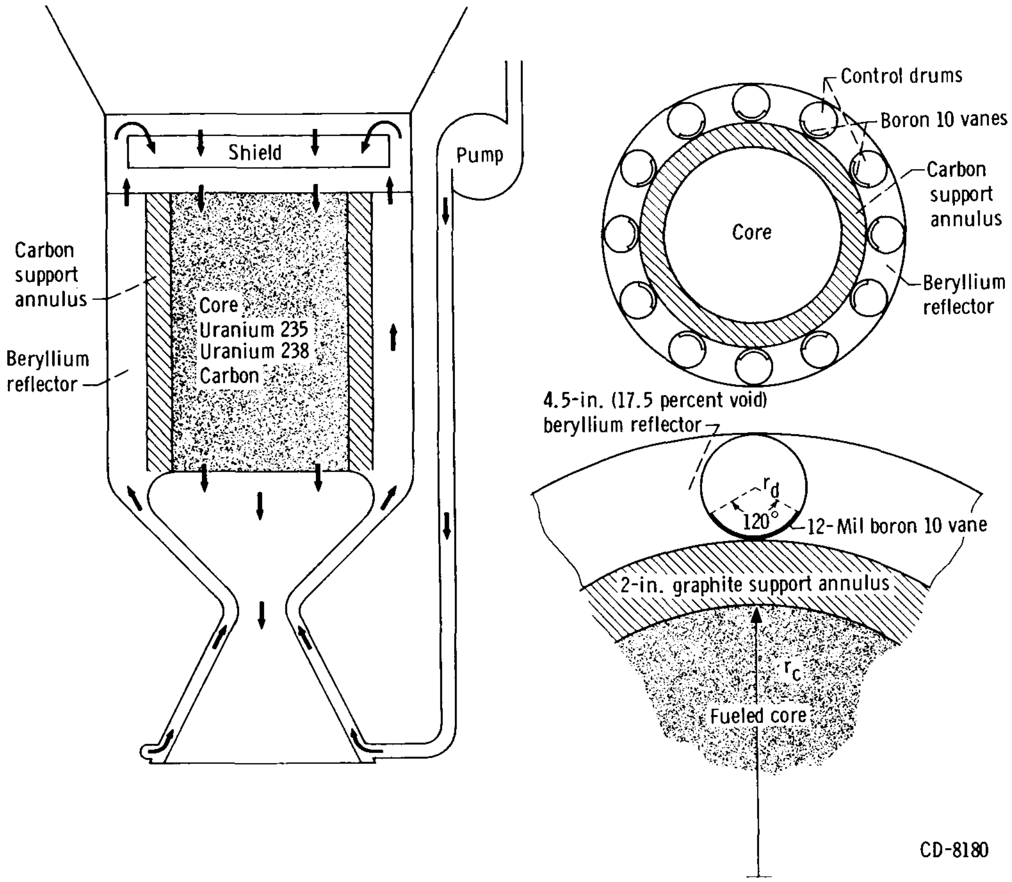

(a) Schematic of reactor system with plan view of reactor showing twelve control drums.

(b) Detailed view of one control drum configuration.

Figure 1. - Nuclear rocket engine.

(Symbols are defined in appendix A.) With the reactors all exhibiting a common effective multiplication factor, the second step in the analysis, the determination of drum span, could then be undertaken.

Figure 2 shows the reactor representation used for the one-dimensional calculation of drum span. A solid cylindrical sheath, 12 mil thick, of boron 10 was placed between the carbon support annulus and the beryllium reflector (drum-in position) and a value of K_{eff} calculated. The boron sheath was then removed from this inner position and placed at the outermost edge of the beryllium reflector and again an effective multiplication factor was calculated. The resulting difference in the K_{eff} values is a measure of the total annular sheath worth. Once this total worth is known, it is necessary to devise a geometrical relation for the annular sheath that will adequately describe the true control vane area. For this study a modified version of the calculational procedure for control vane worth as used by A. W. Charmatz of LASL (ref. 1) was adopted. For a detailed description of this calculation, see appendix B.

Three computer codes were used for this study: a diffusion code, a broad fast group

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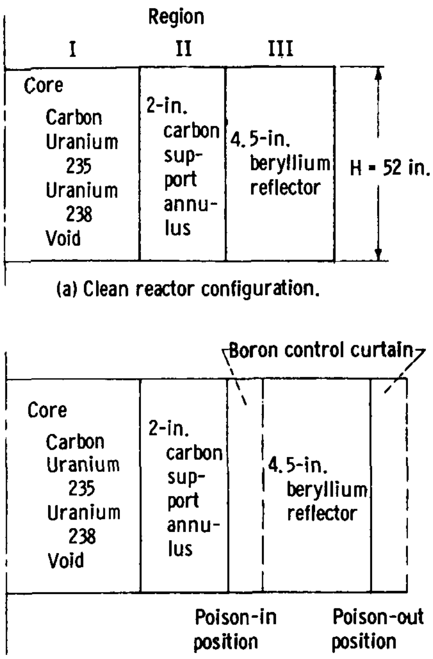

(b) Reactor configuration with boron control curtains.

Figure 2. - Reactor mathematical model one-dimensional analysis.

cross section code, and a thermal group cross section code. A brief description of these codes follows.

Diffusion Code

Spatial calculations were performed using a general one-dimensional, multigroup, multiregion diffusion code. The code was developed at Lewis specifically for the Lewis Research Center's IBM 7094 computer system. The code henceforth will be referred to as RP-1. The equation solved for each energy group by the code, is of the form

$$-D_K \nabla^2 \varphi_K(r) + \Sigma_K \varphi_K(r) = \frac{\alpha_K}{K_{eff}} P(r) + \sum_{\substack{J=1 \\ (J \neq K)}}^{NG} \Sigma(J \rightarrow K) \varphi_J(r)$$

Two features of this code are that (1) the extrapolation distance at the outer boundary is group dependent and that (2) variable transverse buckling was used for each energy group.

Cross Section Codes

Two nuclear cross section codes were used to generate the cross sections required for the solution of the diffusion equation. The broad fast group cross sections were calculated by the General Atomics proprietary code GAM II (ref. 3), while the thermal group constants were determined by using the Atomics International code Tempest (ref. 4).

The GAM II code was used to obtain spectrum averaged multigroup cross sections for each region of the reactor. A uranium 235 fission source spectrum was used in an infinite homogenized media representing each reactor region.

The Tempest code computes the neutron energy spectrum based on either the Wigner-Wilkins light moderator equation, the Wilkins heavy moderator equation, or the formula for the Maxwellian distribution. The code provides macroscopic and microscopic cross sections averaged over the computed thermal neutron spectrum. As in GAM II, the cross sections are generated for an infinite homogeneous media representing each region. The thermal group spectrum covers the energy range from 0.0 to 0.414 electron volts, the GAM II energy minimum.

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TABLE I. - NEUTRON ENERGY SPLITS

Group	Energy, eV	Lethargy
1	14.9×10^{6} to 2.231×10^{6}	-4.0×10^{-1} to 1.50
2	2.231×10^{6} to 8.2085×10^{5}	1.50 to 2.50
3	8.2085×10^{5} to 5.53085×10^{3}	2.50 to 7.50
4	5.53085×10^{3} to 6.1442×10^{1}	7.50 to 1.20×10^{1}
5	6.1442×10^{1} to 0.414	1.20×10^{1} to 1.70×10^{1}
6 (thermal)	0.414 to 0.0	1.70×10^{1} to ∞

In this study six energy groups were used, five fast and one thermal. For a breakdown of the energy splits see table I. A description of the calculational procedure used to determine atom densities for these cross section codes is presented in appendix C.

Comparison of Calculations

Calculations were first made for graphite moderated unreflected spherical cores at room temperature conditions to determine the critical (K_{eff} = 1.00) core radii. The results of these calculations were then compared with the experimental data recorded by the University of California Radiation Laboratory (UCRL (ref. 5)) and the analytical results reported by C. B. Mills (LASL (ref. 6)) and R. K. Plebuck of Massachusetts Institute of Technology (MIT (ref. 7)). The experimental results are shown in figure 3, marked U-1, U-2, U-3, and U-4. C. B. Mills used an 18-group diffusion code with selected nuclear parameters taken from a survey of experimental cores to determine the critical diameters of a host of graphite moderated unreflected spheres ranging in moderator to fuel ratios of 1 to 10 000. Since Mills' curve utilizes the best experimental data available, it will be used as the principal reference for this report.

The results of a 6-group diffusion calculation are compared with Mills' 18-group diffusion calculation and the experimental UCRL results in figure 3. The agreement seems fairly good, with the largest disagreement occurring at a moderator to fuel ratio of about 200. At this point, a deviation of about 3 centimeters is observed.

For the 6-group diffusion results it was noted that, in the range of moderator to fuel ratio of 300 to 1000, the critical radius for the unreflected sphere actually became slightly smaller as the moderator to fuel ratio increased. This same anomoly was observed in the reflected cylinder analysis also. It is felt that this effect can probably be attributed to the limitations on the GAM II multigroup cross section code.

The uranium 235 cross sections are in histogram form to represent the resonance absorption and fission cross sections. The GAM II code averages the absorption and

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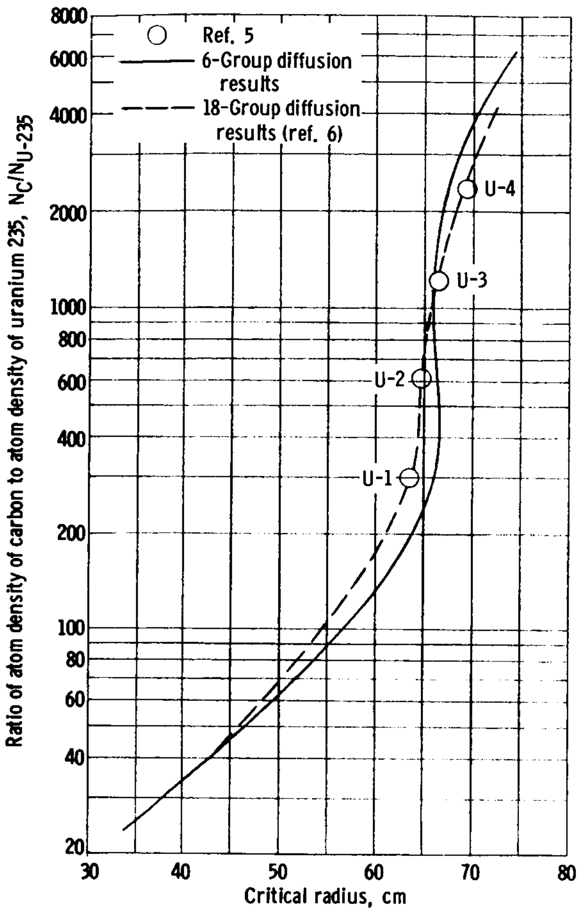

Figure 3. - Critical radii for graphite moderated unreflected spherical cores of varying moderator to fuel ratio.

fission cross sections over 99 broad groups. In the moderator to fuel ratio range of 300 to 1000 where the reactors' mean fission energy passes from fast to epithermal and through the resonance region of uranium 235, the critical diameters are very sensitive to the cross section values. This rather broad averaging of these cross section values (step functions) in this resonance region by GAM II causes the critical diameters to become smaller as the moderator to fuel ratio increases to about 1000. More accurate results in this region of moderator to fuel ratio can be realized by utilizing a cross section code with many more energy groups (approaching 1000).

For further comparison, calculations were carried out to determine the critical diameter of actual experimental rocket reactors, namely, Kiwi and NRX. Since the values of the critical core diameter for the Kiwi and NRX reactors are classified, only the percent agreement can be reported. On the critical core diameter of Kiwi B-4B the diameter was calculated to within 1 percent of the actual value (ref. 1). For the NRX core, the nuclear densities were volume averaged to give a one-region homogenized core with the ensuing calculation yielding a critical diameter within 1.5 percent of the correct value (ref. 2).

Of prime interest to this report is the determination of peripheral control worth. For the cold (room temperature), clean (no nuclear poisons) homogenized Kiwi reactor, a control drum span worth was calculated to within 5 percent of the accepted experimental value. The method used to compute the drum span worth will be discussed in detail in appendix B.

GENERAL DISCUSSION

Because future nuclear rocket engines may well grow in size, primarily radially, the problem of determining the limits to which peripheral control is effective is of major

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concern. The effectiveness of a control drum system depends on the span worth of the drums. This, as discussed earlier, is measured by the difference in reactivity of a reactor for the drums turned to a poison-in position and the drums in a poison-out position.

Each reactor configuration investigated is one having a core size corresponding to $K_{eff} = 1.050$ with no poison in the reflector and the core is cold (room temperature) and clean (no nuclear poisons). It is felt that this "critical" ($K_{eff} = 1.050$) condition would provide an adequate margin of reactivity for subsequent introduction of core structure, poisoned control drums, and final shimming. In all the cases examined, the core length was finite with no end reflection.

The effect of various reactor parameters on span worth was investigated to determine peripherial control drum characteristics. The parameters studied were core moderator to fuel ratio R, core void fraction α, core height H, reflector thickness, reflector void, and composition of the support annulus. For each set of parameters the core diameter was first determined and then the span worth of the control drums evaluated.

Determination of Core Diameters as Function of Core Void Fraction and Moderator to Fuel Ratio for $K_{eff} = 1.050$

To show the effect of core void fraction α and core moderator to fuel ratio R, core diameters corresponding to a range of core voids from 20 percent to 40 percent and R values of 100 to 3000 were evaluated and the results presented in table II. Figure 4 shows the relation between the core diameter, the moderator to fuel ratio, and the core void for the following set of fixed reactor parameters: 4.5-inch beryllium reflector, 2-inch carbon support annulus, and 52-inch core height.

In the cases examined, an increase in the core void fraction for a particular moderator to fuel ratio caused an increase in the core diameter to establish the required critical condition. The axial neutron leakage for these assemblies almost doubles as the void fraction is doubled. As the moderator to fuel ratios and the void fraction increase, the significance of this void change becomes greater and greater. At about 40 percent void, the curves flatten out so that the core diameter must undergo an enormous change in order to maintain a constant K_{eff}. This is the result of the increasing importance of thermal leakage. Since the moderator to fuel ratio defines the neutron energy spectrum and since the moderator to fuel ratio is increasing, pushing the reactor more and more toward the thermal range, the influence of the thermal leakage becomes more predominant. Thus the core must increase in size at an ever faster rate to compensate for the leakage and maintain a constant effective multiplication factor. The core diameters shown in table II refer to the diameter of the fueled region only.

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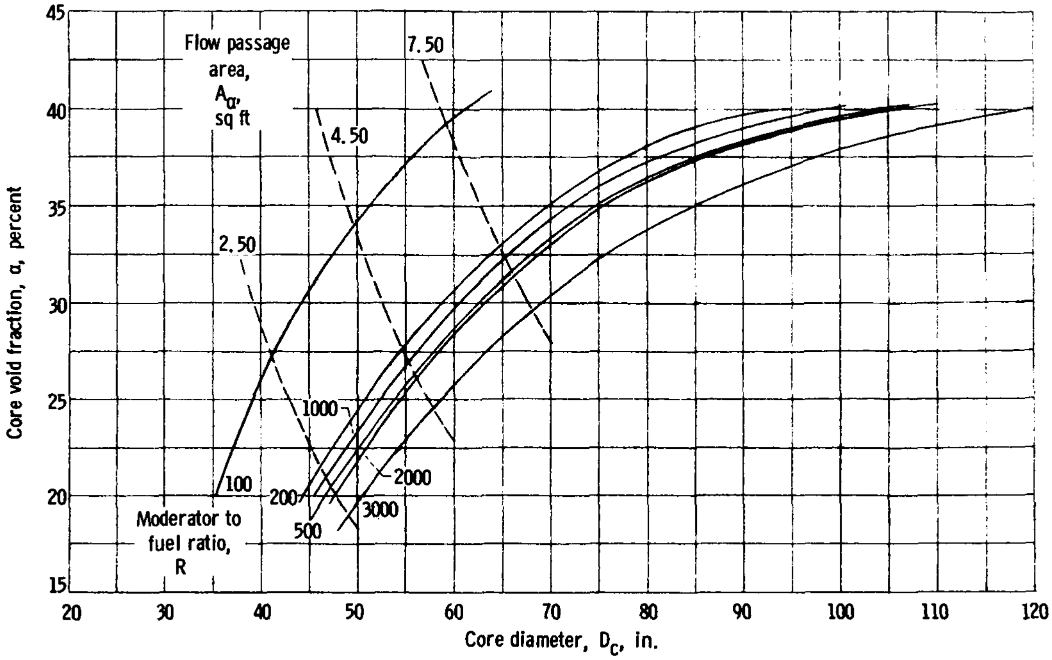

Figure 4. - Core diameters for specific moderator to fuel ratios and various void fractions such that static criticality factor K_{eff} = 1.050. Reactor configuration: 4.5-inch (17.5 percent void) beryllium reflector; 2-inch carbon support annulus; core height, 52 inches.

TABLE II. - CORE DIAMETER FOR SPECIFIC MODERATOR TO FUEL RATIOS AND VARIOUS CORE VOID FRACTIONS SUCH THAT K_{eff} = 1.050

Core void fraction, α, percent	Moderator to fuel ratio, R						
	50	100	200	500	1000	2000	3000
	Core diameter, D_c, in.						
20	24.85	35.20	43.78	46.51	45.65	47.74	50.60
30	29.63	44.11	57.67	62.48	60.85	64.03	68.98
40	37.72	61.38	91.118	104.65	99.54	106.43	120.71

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The dashed cross plots found in figure 4 are curves of constant flow passage area. The flow passage area is defined as cross-sectional area times the core void fraction, that is,

$$A_\alpha = \frac{\pi D_c^2}{4} \alpha$$

where D_c is the diameter of the core (fueled region) measured in feet.

Since this parameter can be related to power output of the reactor, it will be of value to determine how the control span varies for constant flow passage area as the moderator to fuel ratio and core diameter increases. This will be discussed in the following section (determination of peripheral drum worth) of this report. The drum span worth for this set of reactor configurations is shown in figure 5. In all the cases examined, there was a significant loss of control (drum span) as the core diameters increased.

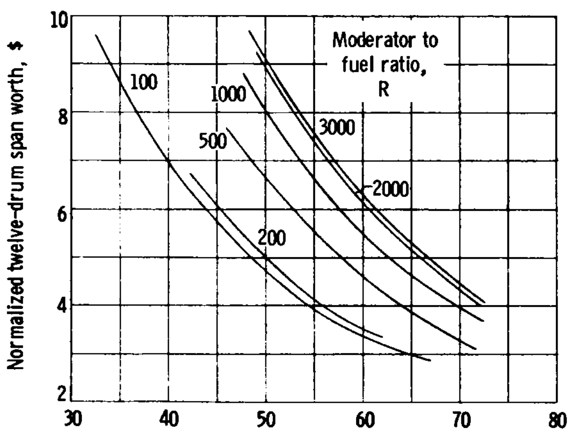

Figure 5. - Control span for reactor cores of varying moderator to fuel ratio.

Figure 6 shows the span worths for cores of constant flow passage area and varying moderator to fuel ratio. It is of interest to note that for a particular flow passage area there is a point in the growth of the core where a sharp increase in controllability can be realized by increasing the moderator to fuel ratio.

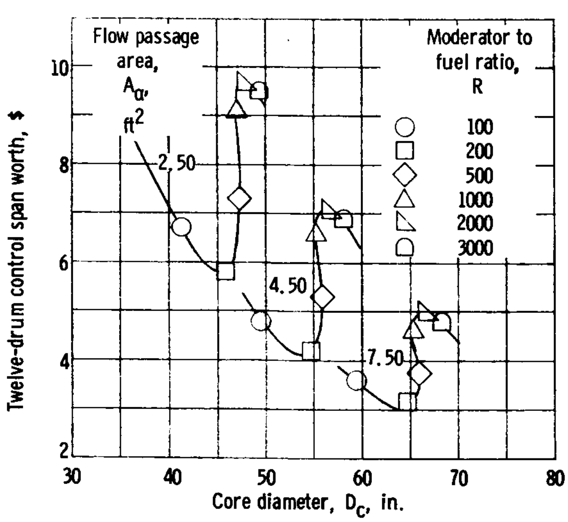

Figure 6. - Control span worths for cores of constant flow passage area and varying moderator to fuel ratio.

Determination of Drum Span Worth for Reactors of Varying Height

To show the effect of reactor height H, core diameters were evaluated for a fixed set of reactor parameters, that is, R = 100, 2-inch carbon support annulus, and a 4.5-inch beryllium reflector over a range of core voids from 20 percent to 40 percent and core heights from 52 to 60 inches.

Figure 7 shows the core diameters for the range of core voids and core heights

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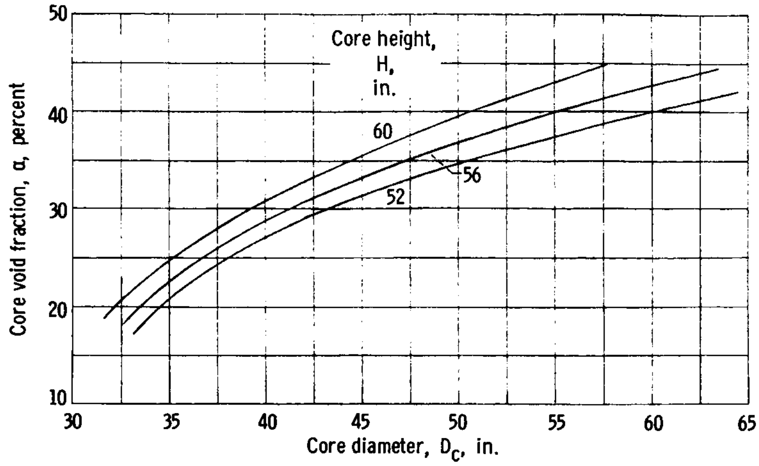

Figure 7. - Core diameters for reactors of varying height and changing core void fraction such that static criticality factor $K_{eff} = 1.050$. Reactor configuration: 4.5-inch (17.5 percent void) beryllium reflector; 2-inch carbon support annulus; moderator to fuel ratio, 100.

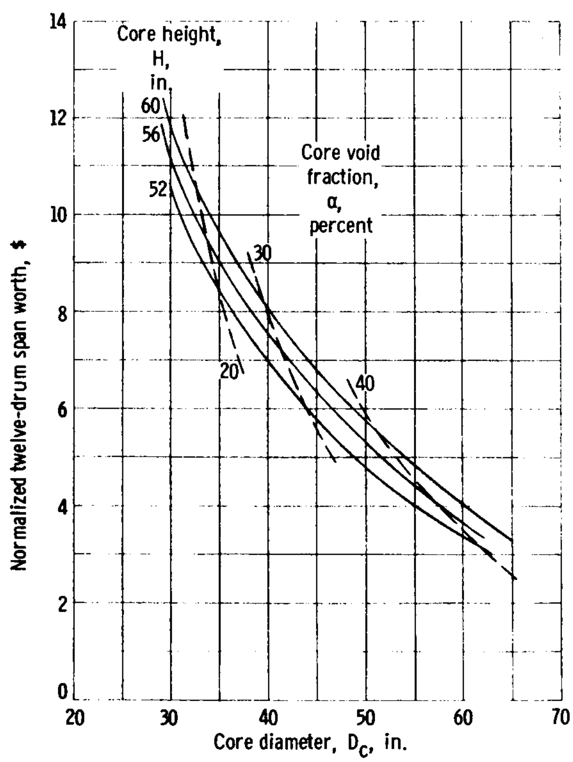

Figure 8. - Control span for cores of varying height. Reactor configuration: fueled core, carbon-uranium 235-uranium 238; 4.5-inch (17.5 percent void) beryllium reflector; 2-inch carbon support annulus; moderator to fuel ratio, 100.

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examined. The curves of figure 7 show that it is possible to maintain the same core diameter but increase the flow passage area (increase in core void) by simply increasing the core height. At the larger core diameters of 55 or 60 inches, the increase in core height will permit a larger increase in core void and still maintain a reactor K_{eff} equal to 1.050.

Span worths for this set of reactor configurations are shown in figure 8. From these plots it is evident that almost one dollar can be gained in control span worth by increasing the reactor height from 52 to 60 inches. The dashed curves of figure 8 are the core void fractions. By utilizing the core void fraction curves, trade offs in core diameter and core void can be made.

Determination of Drum Span for Reactors of Varying Reflector Thickness

To show the effect of increasing the beryllium reflector thickness on core diameter and drum span, calculations were made for fixed values of the reactor parameters, that is, R = 100, H = 52 inches, 2-inch support annulus, and a reflector void of 17.5 percent. Figure 9 shows the core diameter as a function of beryllium reflector thickness and core void for voids ranging from 20 to 40 percent and reflector thicknesses from 4.5 to 12.5 inches. Examination of figure 9 reveals that there is a practical limit on reflector size beyond which no appreciable effect on core diameter is achieved. Also shown in figure 9 are the reflector savings curves for this same set of reactor configurations.

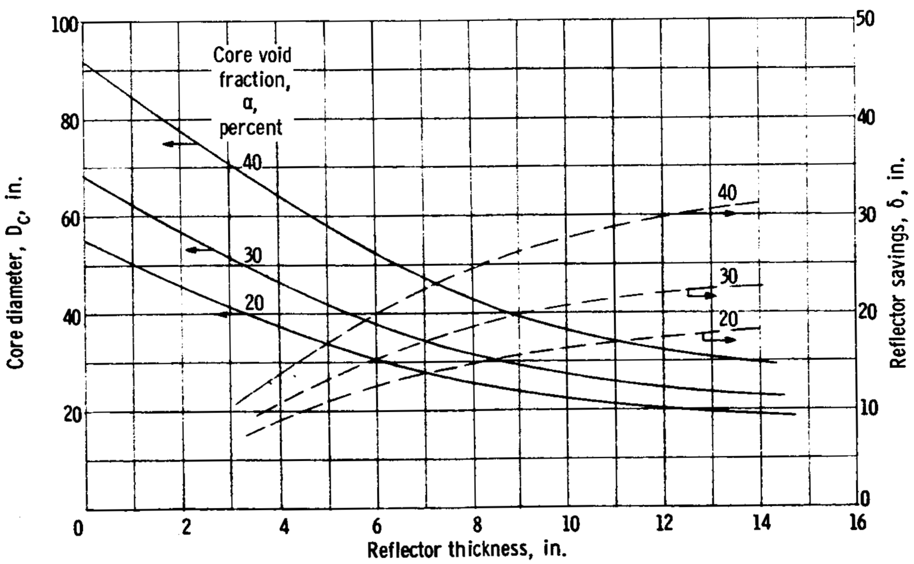

Figure 9. - Core diameters and reflector savings for reactors of varying reflector thickness. Reactor configuration: 4.5-inch (17.5 percent void) beryllium reflector; 2-inch carbon support annulus; static criticality factor K_{eff}, 1.050; core height, 52 inches; moderator to fuel ratio, 100.

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TABLE III. - CORE SIZES FOR SEVERAL REFLECTOR THICKNESSES AND CORE VOIDS

[Static criticality factor $K_{eff} = 1.05$.]

Reflector thickness, in.	Core void fraction, α, percent		
	20	30	40
	Core diameter, D_c, in.		
0.0	55.28	68.38	91.79
4.5	35.20	44.11	61.39
8.5	24.58	30.14	40.94
12.5	20.00	23.99	31.44

TABLE IV. - REFLECTOR SAVINGS FOR SEVERAL REFLECTOR THICKNESSES AND CORE VOIDS

[Static criticality factor $K_{eff} = 1.05$.]

Reflector thickness, in.	Core void fraction, α, percent		
	20	30	40
	Reflector savings, δ, in.		
4.5	10.04	12.13	15.20
8.5	15.35	19.12	25.43
12.5	17.64	22.20	30.18

Reflector savings is defined as (ref. 8)

$$\delta = r_{c,bare} - r_{c,refl}$$

where

$r_{c,bare}$ radius of bare core, in.

$r_{c,refl}$ radius of core with reflector, in.[1]

Tables III and IV show the core diameters and reflector savings for the range of core voids and reflector thicknesses examined.

Drum span for varying core diameters. - Drum span worths for this set of reactor configurations (up to 8.5-in. reflectors) are shown in figure 10. These curves show that the drum span increases rapidly with increasing reflector thickness. The bulk of the drum span for the thicker reflected cores stem from the severe depression of K_{eff} ($\ll 1.000$) when the boron sheath is inserted in the inner position. Since an adequate margin for shutdown for this type of reactor is of the order of 10 dollars and approximately 7 dollars in excess is available from these cores, a drum span greater than 20 or 25 dollars would be superfluous. Curve A and a portion of curve B may therefore be unnecessary, but they were plotted, nevertheless, to show the strong influence on drum span generated by increasing the reflector thickness.

Because the reflector thickness was found to be so important to drum span, a second study of this parameter was made.

Drum span for constant core diameters. - This drum span analysis is based on the assumption that an increase in excess reactivity is demanded of a particular reactor configuration. By holding the core dimensions and core void constant, this excess can be obtained by increasing the thickness of the beryllium reflector.

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[1] The reflector in this analysis includes a constant 2-inch carbon support annulus and the beryllium reflector.

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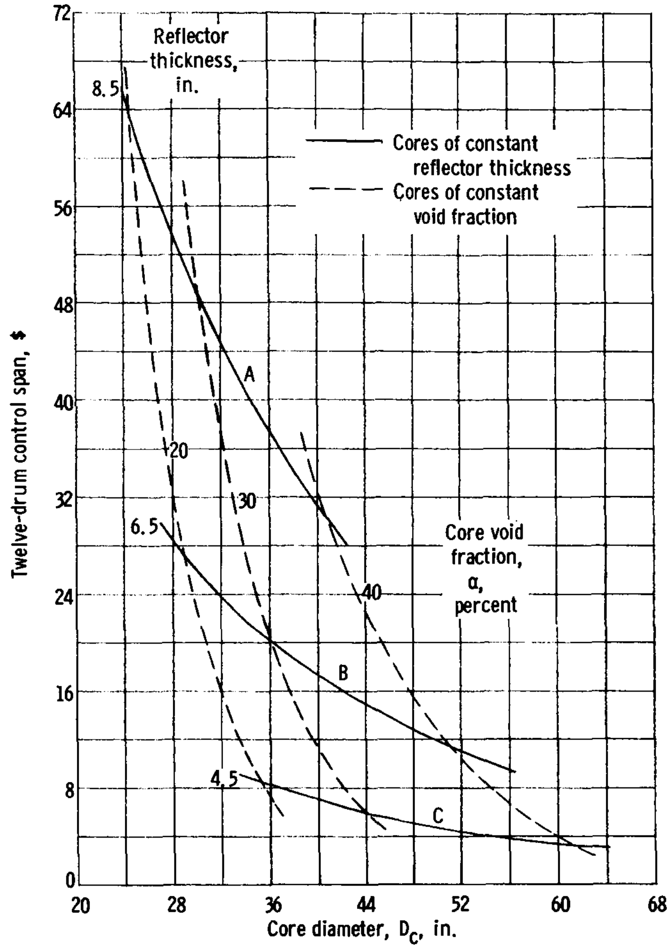

Figure 10. - Control drum span for cores of varying reflector thickness. Reactor configuration: 17.5 percent void in beryllium reflector; 2-inch carbon support annulus; core height, 52 inches; moderator to fuel ratio, 100.

The three reactor assemblies chosen for this study were the following:

(1) Core diameter D_1 = 35.20 inches, core void fraction α = 20 percent

(2) Core diameter D_2 = 44.11 inches, core void fraction α = 30 percent

(3) Core diameter D_3 = 61.38 inches, core void fraction α = 40 percent

To show the effect of increasing reflector thickness on drum span for the reactor assemblies just described, the beryllium reflector, with a 17.5 percent void, was increased from 4.5 to 12.5 inches. Figure 11 shows the drum span for these reactor configurations. Here again, the drum span increases rather rapidly with increasing reflector thickness, thereby reaffirming the importance of the reflector on drum span.

Determination of Drum Span Worth for Reactors of Varying Reflector Void

To show the effect of varying reflector void, calculations were made at fixed values

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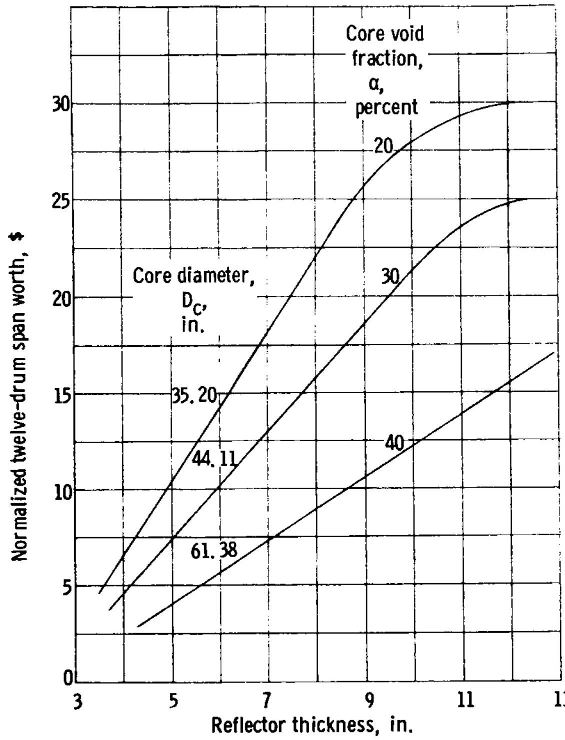

Figure 11. - Control span for three cores of constant diameter and varying reflector thickness. Reactor configuration: no top or bottom reflectors; 2-inch support annulus; core height, 52 inches; moderator to fuel ratio, 100; room temperature conditions.

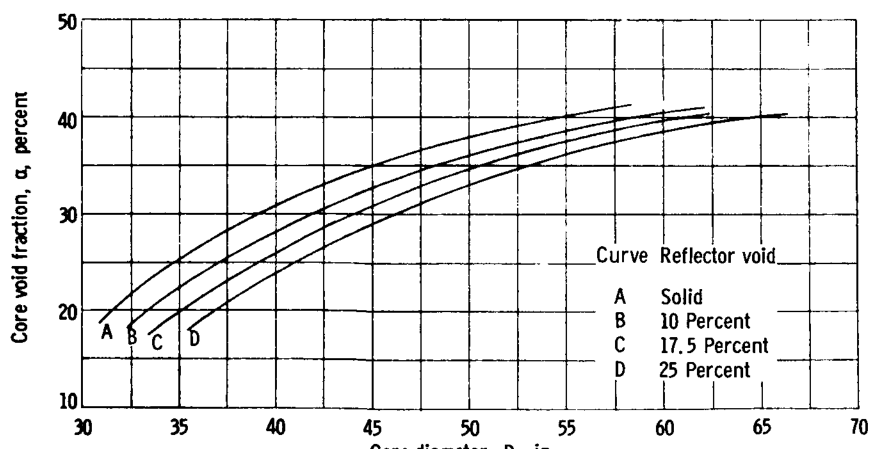

Figure 12. - Core diameters for constant beryllium reflector thickness with varying reflector void. Reactor configuration: 4.5-inch beryllium reflector; 2-inch carbon support annulus; static criticality factor K_{eff}, 1.050; core height, 52 inches; moderator to fuel ratio, 100.

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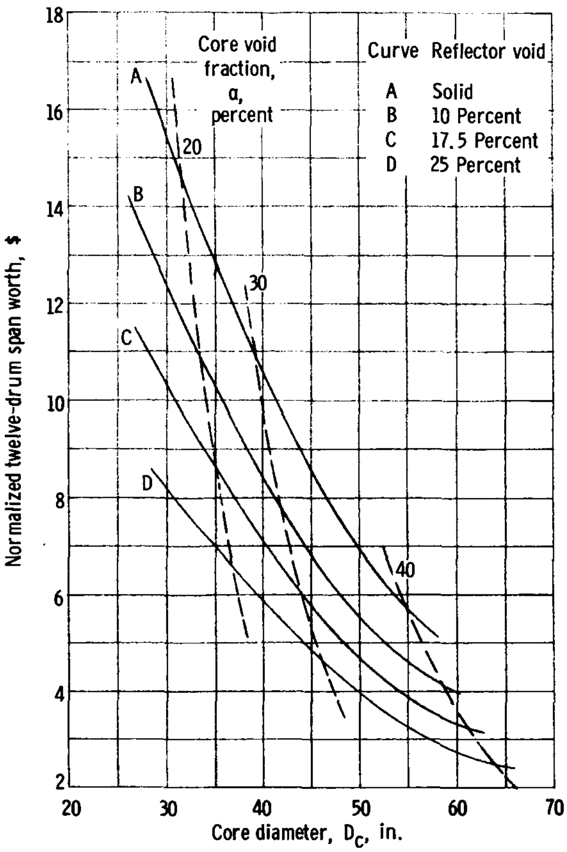

Figure 13. - Control span for large cores of varying reflector void.

of core moderator to fuel ratio (R = 100), core void fractions (20, 30, and 40 percent), core height (52 in.), reflector thickness (4.5 in.), and a 2-inch carbon support annulus. Figure 12 shows the core diameters for a range of reflector voids from solid to 25 percent.

Figure 13 shows the normalized drum span worth for this family of reactor configurations. The dashed cross plots show the core voids of 20, 30, and 40 percent.

Determination of Drum Span Worth for Reactors with an Aluminum Support Annulus

To this point in the parametric study, the carbon support annulus has remained a constant item in the reactor configurations analyzed. Even as the reactor cores grew to rather large diameters (>50 in.) the carbon annulus was retained. Since it is difficult to manufacture carbon annuli of large diameters and because the carbon may not be mechanically adequate at these large core sizes, it will be necessary to examine an alternate material for the support annulus.

Because of its rather good neutronic characteristics and its structural capabilities, aluminum was chosen as the element to replace the carbon.

From the section Determination of Core Diameters as Function of Core Void, three reactor models were chosen. The assemblies had the appropriate core diameter, core fuel mixture, height, and reflector-support annulus configurations to give an effective multiplication factor of 1.050 (fig. 2, p. 4). The reactor models used for this analysis have the moderator to fuel ratio R = 100 with core void of 20, 30, and 40 percent, a 2-inch carbon support annulus and a 4.5-inch beryllium reflector with a 17.5 percent void.

To determine the worth of the aluminum support annulus to that of a carbon one, the carbon annulus was removed from the aforementioned reactor models and an aluminum sheath, of equal thickness, was inserted. The resulting calculation of the effective multiplication factor was a measure of the worth of the aluminum annulus to the reactor. As was expected, the K_{eff} for the aluminum assemblies dropped in value. The results are

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TABLE V. - WORTH OF ALUMINUM SUPPORT ANNULUS FOR CORES OF CONSTANT MODERATOR TO FUEL RATIO (R = 100) AND VARYING CORE VOIDS

Critical diameter, in.	Core void fraction, α, percent	Worth of aluminum support annulus, $
35.20	20	8.63
44.11	30	7.43
61.38	40	5.49

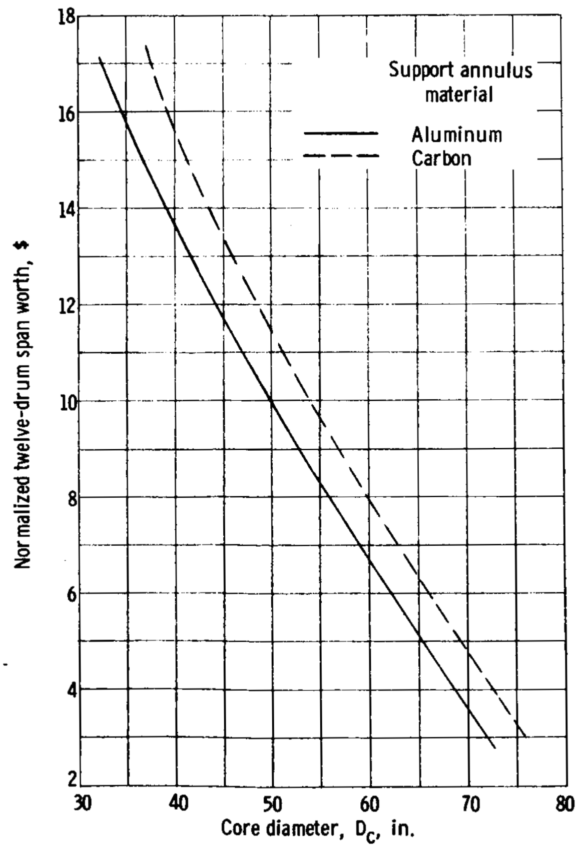

Figure 14. - Control span for reactors with 7.25-inch beryllium reflector, 2-inch support annulus, core height of 52 inches, and moderator to fuel ratio of 100.

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shown in table V.

Next a calculation was made to determine the increase in thickness of the beryllium reflector necessary to make the core aluminum support annulus configuration have a K_{eff} equal to 1.050. In the three cases examined (core voids of 20, 30, and 40 percent), the beryllium reflector had to be increased to approximately 7.25 inches in order to have the required excess reactivity.

Once the correct reflector thickness has been established, the boron 10 control sheath was inserted and a drum span calculation conducted. The results of these calculations are shown in figure 14 (solid curve). Also plotted in figure 14 is the comparable carbon support annulus configuration. Examination of the two curves reveals that, for the larger cores, greater than 50 inches in diameter, approximately $1.20 of drum span is lost with the use of the aluminum support annulus.

CONCLUDING REMARKS

A general one-dimensional, multigroup, multiregion diffusion code written for the IBM 7094 was used in this study to define more clearly the characteristics of a peripheral control system for graphite moderated reactors. The program computed critical core sizes and drum span worth for a broad spectrum of reactor configurations.

The calculations of the critical core size for spherical unreflected graphite-uranium 235 of varying moderator to fuel ratios agree rather well with the experimental data reported by C. B. Mills of Los Alamos and the theoretical results published by R. K. Plebuck. The subsequent analysis of core diameters for cylindrical reflected graphite-uranium 235 of varying moderator to fuel ratios exhibited the same general characteristics as those for the spherical unreflected cores.

In the control of graphite moderated cores from a peripheral system, it was observed from the calculations that control worth would decline with increasing core diameter at a constant moderator to fuel ratio. An increase in peripheral control was realized by (1) increasing the surrounding beryllium reflector thickness and (2) increasing the core height at any one core diameter.

A change of the core support structure from carbon to aluminum decreased the effective control span worth. Core void increases and changes to the core support structure can be compensated for by increasing the thickness of the beryllium reflector. However, there appears to be a limit to the gain in control that can be realized by increasing the reflector thickness.

The investigation described in this report was meant to show the control characteristics or trends resulting from the perturbations of various reactor parameters and not precise design limits. More detailed calculations would certainly be necessary for the

[redacted]

actual design of a specific large core rocket reactor. Such calculations should consider in detail two-dimensional calculations to more nearly simulate the control geometry, a more realistic reactor configuration that would include core structural material, hydrogen and reactor shields, spatial effects on cross sections, an accurate calculation of the uranium 235 resonance cross sections, up and down scattering, and the effective delayed neutron fraction calculation.

Lewis Research Center,
National Aeronautics and Space Administration,
Cleveland, Ohio, September 28, 1965.

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APPENDIX A

SYMBOLS

A atomic number

A_α flow passage area (surface area core void), ft^2

D_c diameter of core (fueled region), in.

D_K diffusion coefficient, cm

H core height, in.

K_{eff} static criticality factor

K_1 effective multiplication factor for clean (no control poison) reactor configuration, 1.050

K_2 effective multiplication factor for reactor configuration with control sheath inserted

m geometry indicator (0-slab, 1-cylinder, 2-sphere)

N atom density, atoms/(b)(cm)

NG total number of groups

N_o Avogadro's number

$P(r)$ total neutron production at r, neutrons/(cm^3)(sec)

R moderator to fuel ratio, $N_C(\text{total})/N_{U-235}$

R^* $N_C(\text{total})/N_{UC_2}$

r_c radius of core, in.

$r_{c,bare}$ radius of bare core, in.

$r_{c,refl}$ radius of core with reflector, in.

r_d radius of control drum, in.

t_s support annulus thickness, in.

V core fraction

α core void fraction, percent

α_K fraction of source neutrons born in any particular energy group

β_{eff} effective delayed neutron fraction, 0.0075

∇^2 one-dimensional Laplacian, $\dfrac{d^2}{dr^2} + \dfrac{m}{r}\,\dfrac{d}{dr}$

δ reflector savings, in.

$\delta\rho$ difference in reactivity between two calculated eigenvalues K_1 and K_2, \$

ρ material density, g/cm^3

$\Sigma(J \rightarrow K)$ transfer cross section from group J to K

Σ_K total loss operator

$\varphi_K(r)$ flux at r energy group K, neutrons/(cm^2)(sec)

Subscripts:

C carbon

U-235 uranium 235

U-238 uranium 238

UC_2 uranium dicarbide

[redacted]

[redacted]

APPENDIX B

DETERMINATION OF DRUM SPAN FOR TWELVE-DRUM PERIPHERAL CONTROL SYSTEM

With the core diameter established for a given reactor configuration, such that the $K_{eff} = 1.050$, the boron control sheath was introduced, first in the "in-position", then in the "out-position", as shown in figure 2 (p. 4). A K_{eff} was calculated for each of these reactor assemblies and the results normalized according to the following relation (ref. 8):

$$\delta\rho = \frac{K_1 - K_2}{K_1 K_2 \beta_{eff}}$$

This normalized difference in reactivity is then a measure of the total cylindrical control sheath worth.

Calculations with the control sheath in the out-position (fig. 2, p. 4) yield little or no difference in the effective multiplication value from the clean, no control reactor configuration. The normalized sheath worth in the out-position does not significantly contribute to the drum span and therefore will be omitted from the calculation.

The control sheath worth must now be divided in a manner proportional to the ratio of the actual control vane area and the total curtain area. A. W. Charmatz of LASL (ref. 1) has devised a relation that describes the effective neutron absorbing area for a single control drum vane. This relation will be adopted for this report.

The twelve-drum span worth is calculated as follows:

$$\frac{\text{effective control vane area}}{\text{total sheath area}} \times \text{total curtain worth (\$)} \times 12 \text{ drums} = \text{twelve-drum span worth (\$)}$$

The effective control vane area is

$$\frac{1}{3}\, 2\pi(r_d - 0.59 \text{ in.})H$$

where 0.59 inch is the approximate thermal neutron mean free path in beryllium.

The total sheath area is

$$2\pi(r_c + t_s)H$$

and the total sheath worth in dollars is

$$\frac{K_1 - K_2}{K_1 K_2 \beta_{eff}}$$

[redacted]

[redacted]

APPENDIX C

DETERMINATION OF ATOM DENSITIES FOR VARIOUS CARBON TO URANIUM FUEL RATIOS

Let

$$V_C + V_{UC_2} + \alpha = 1 \tag{C1}$$

Let

$$R = \frac{N_C(\text{total})}{N_{U\text{-}235}} \tag{C2}$$

Let

$$R^* = \frac{N_C(\text{total})}{N_{UC_2}} \tag{C3}$$

The total number of carbon atoms is made up of two factors: (1) the main core carbon and (2) the carbon in the uranium dicarbide. Thus,

$$R^* = \frac{N_{C,\text{core}} + 2N_{C,UC_2}}{N_{UC_2}}$$

$$= \frac{N_{C,\text{core}}}{N_{UC_2}} + 2$$

$$= \frac{V_C \rho_C A_{UC_2}}{V_{UC_2} \rho_{UC_2} A_C} + 2$$

If $\rho_C = 1.60$ grams per cubic centimeter, $A_{UC_2} = 259.23$, $\rho_{UC_2} = 11.68$ grams per cubic centimeter, and $A_C = 12.011$, then

$$R^* = 2.976 \frac{V_C}{V_{UC_2}} + 2$$

[redacted]

[redacted]

$$V_{UC_2} = \frac{2.976V_C}{R^* - 2}$$

$$V_C + \frac{2.976V_C}{R^* - 2} = 1 - \alpha$$

$$V_C = \frac{(R^* - 2)(1 - \alpha)}{R^* + 0.976}$$

Now

$$N_C(\text{total}) = N_C + 2N_{UC_2}$$

$$N_C(\text{total}) = \frac{V_C \rho_C N_o}{A_C} + 2\,\frac{V_{UC_2} \rho_{UC_2} N_o}{A_{UC_2}}$$

$$N_C = 0.0803V_C + 0.0539V_{UC_2}$$

$$N_C = 0.0803\,\frac{(R^* - 2)(1 - \alpha)}{R^* + 0.976} + 0.0539\,\frac{2.976}{R^* - 2}\,\frac{(R^* - 2)(1 - \alpha)}{R^* + 0.976}$$

$$N_C = \frac{1 - \alpha}{R^* + 0.976}\left[0.0803(R^* - 2) + 0.1604\right]$$

$$N_{UC_2} = \frac{N_C}{R^*} \quad \text{and} \quad N_{U\text{-}235} = 0.93N_{UC_2}$$

$$N_{U\text{-}235} = \frac{0.93N_C}{R^*}$$

$$N_{U\text{-}238} = \frac{0.07N_C}{R^*}$$

$$R^* = 0.93R$$

The uranium fuel is assumed to be 93 percent uranium 235 and 7 percent uranium 238.

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